UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, et al.,	)	Docket Nos. 50-443 50-444
(Seabrook Station, Units 1 and 2)	)	

NECNP SUPPLEMENTAL PETITION FOR LEAVE TO INTERVENE

Pursuant to 10 CFR 2.714(b), the New England Coalition on Nuclear Pollution submits the following contentions for litigation in this proceeding. They are based on information available to date in the FSAR and the files of the NRC Public Document Room. To a degree, their development depended upon the extent to which the Applicant submitted an FSAR that was coherent and could be followed clearly. We discovered instances in which issues that should have been discussed in one section were covered in others and various other instances in which the FSAR was difficult to follow. We have attempted to ferret out as many of these hidden points as possible. We will continue to do so and will file additional contentions where warranted.

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As a clarifying point, we have relied in several instances on Regulatory Guides as the basis for contentions. Our point is not that Regulatory Guides constitute NRC requirements, but that the Regulatory Guides themselves constitute factual bases for our contentions. In particular, they provide a benchmark against which the Board may judge compliance with the regulations.

Respectfully submitted,

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Dated: April 21, 1982

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Contentions Index

I. TECHNICAL SAFETY CONTENTIONS

A. Environmental Qualification - Electrical Equipment

1. Environmental qualification, or the requirement that all safety systems must be able to function in the accident environment, is "fundamental to NRC regulation of nuclear power reactors." CLI-78-6, 7 NRC 400, 412 (1978). NECNP contends that the Seabrook facility cannot be licensed because it does not meet the Commission's standards for environmental qualification of electrical equipment under 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4. Furthermore, in light of the Three Mile Island accident, GDC 4 requires more rigorous environmental qualification testing than was previously the case in order to provide reasonable assurance that electrical equipment will function for the entire time period in which it is needed. The FSAR's discussion of environmental qualification is deficient in four respects: 1) the parameters of the relevant accident environment have not been identified; 2) the length of time the equipment must operate in the accident environment has not been included as a factor; 3) the methods used to qualify the equipment are not adequate to give reasonable assurance that the equipment will remain operable; and 4) the effects of aging and cumulative radiation exposure on the equipment have not been adequately considered.

Basis:

The Applicant's FSAR states that the Applicant has complied with Regulatory Guide 1.89 in qualifying electrical, instrumentation and control equipment. FSAR at 1.8-33. Regulatory Guide 1.89, however, is not the applicable standard for environmental qualification. The Commission has set DOR Guidelines and NUREG-0588, which are more detailed and specific than Reg. Guide 1.89, as the standard for compliance with GDC 4. CLI-80-21, 11 NRC 521 (1980). The applicant must show compliance with CLI-80-21 in order to obtain an operating license.

A rule has recently been proposed which would extend the compliance deadline from June 30, 1982 (as established in CLI-80-21) to the second refueling outage after that date. 47 Fed. Reg. 2876 (January 20, 1982). However, even under this proposed extension, applicants for operating licenses would be required to submit an analysis ensuring that the plant can be safely operated pending environmental qualification. PR 10 CFR 50.49(k). The applicant has offered no such analysis or assurance that it will be able to meet the current environmental qualification standard.

Furthermore, the accident at Three Mile Island, in which theoretically qualified equipment did not function for the time period in which it was needed, showed that the Commission's standards at that time were inadequate to

provide a reasonable assurance that plants may be operated safely.*/ Although CLI-80-21 was issued after the accident, the Commission stated specifically that it had not attempted to incorporate the lessons of TMI into the decision. In light of the TMI-2 experience, to provide a reasonable assurance that the Seabrook plan can operate safely, the Applicant must show that safety-related equipment can withstand accident conditions for substantially longer than the matter of minutes currently required. The Applicant's environmental qualificiation information is also inadequate to support any finding that environmental qualification is complete because it has completely omitted data on the duration for which it is qualified. It should, therefore, be rejected as insufficient support for the granting of a license. NECNP reserves the right to amend its environmental qualification contention when and if this information is submitted.

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*/Memorandum from S. H. Hanauer, Assistant Director for Plant Systems, DSS, re: Environmental Qualification and Instrumentation (April 6, 1979).

I. A. 2. The Applicant has not complied with Commission standards regarding qualification tests of electric valve operators installed inside the containment.

Basis:

The FSAR indicates environmental qualification of these components is in compliance with IEEE Standards 382-1972 and 323-1974. FSAR at 1.8-28, 29 However, the Applicant must, at a minimum, comply with DOR guidelines and NUREG-0588 as required by the Commission in CLI-80-21 and with the more rigorous environmental qualification testing that the TMI accident demonstrated is necessary to assure safety.

1. A. 3. The Applicant has not complied with GDC 4 in that it has not environmentally qualified electrical equipment inside the containment to withstand the effects of a hydrogen release such as occurred at Three Mile Island Unit 2.

Basis:

The accident at Three Mile Island showed that the accident environment which must be considered in determining whether equipment is environmentally qualified includes the generation of hydrogen gas. At Three Mile Island, hydrogen apparently burned or otherwise disabled electrical equipment inside the containment and thus prevented it from functioning during the accident. Hydrogen burn must therefore be considered a part of the accident environment for purposes of environmental qualification of electrical equipment inside the containment. The Applicant has failed to do so here.

I. B. Environmental Qualification -- Mechanical Equipment

I. The Applicant has not satisfied the requirement of GDC 4 that all equipment important to safety be environmentally qualified to survive and function in the accident environment. In particular, the Applicant does not classify as safety-grade all systems that may be required to remove heat from the steam generators during an accident: e.g., steam dump valves, turbine valves, and the steam dump control system. This omission also violates GDC 5, which requires the Applicant to establish a reliable decay heat removal system.

Basis:

The Three Mile Island accident demonstrated that systems that have not previously been classified as "important to safety" or subjected to environmental qualification requirements may be called upon in accident conditions to mitigate the effect of the accident. NUREG-0578 at 18. There, non-safety grade equipment was used to convey heat away from the core. The Three Mile Island accident prompted a rethinking within the agency of the scope of the environmental qualification "envelope."*/ The Office of Nuclear Reactor Regulation has found that "one of the most important factors in the safety of nuclear reactors is the reliability of the systems used for decay heat removal following the shutdown of the reactor for any reason." NUREG-0705 at A-1. The Three Mile Island accident confirmed that "loss of capability to remove heat through

*/Memorandum from S. H. Hanaver, Assistant Director for Plant Systems, DDS, re: Environmental Qualification and Instrumentation (April 6, 1979).

the steam generator is a significant contributor to the probability of a core meltdown." Id. at A-2. The NRC Staff asserted that "alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and, therefore, could potentially significantly reduce the overall risk to the public." Id. Unless those components which may be called upon to remove residual heat during an accident are environmentally qualified, they cannot be relied upon as part of a reliable system for removing decay heat from the core. The Three Mile Island accident has proved that steam generator heat removal equipment for normal operations must be considered safety-grade.

I. B. 2. The Applicant has not satisfied the requirement of GDC 4 that all equipment important to safety be environmentally qualified because it has not specified the time duration over which the equipment is qualified.

Basis:

Equipment important to safety must be able to withstand the accident environment for the period in which it is required to function. This was made clear by the Three Mile Island accident, during which accident conditions persisted for an unexpectedly lengthy period. The Applicant has not specified in its FSAR what length of time the safety-related mechanical equipment is qualified to operate. It must be required to do so before mechanical equipment can be accepted as environmentally qualified. NECNP reserves the right to challenge the adequacy of the durational parameters if and when this information is submitted to the NRC by the Applicant.

I. C. Environmental Qualification-Emergency Feedwater Pump House HVAC

According to Table 1.3-2, Sheet 14, of the FSAR, the Applicant has added a new heating, ventilating, and air conditioning (HVAC) system for the emergency feedwater pump house. Only parts of the HVAC system are considered safety related and environmentally qualified. Table 3.1(B)-1, Sheet 4. NECNP contends that the entire system and its function must be environmentally qualified, and that the environmental qualification must take into account the likely duration of an accident during which the HVAC system would be relied upon.

Basis:

Since the emergency feedwater pump house and its equipment are capable of functioning and can be relied upon to function only within a particular temperature range, the HVAC system is required to maintain conditions within that range. Accordingly, it must be environmentally qualified to assure that it will be able to function when needed. The environmental qualification must take into account the fact that the equipment may be required to operate for a considerable length of time in the event of an accident.

I. D. Testing of Equipment

1. The Applicant has not complied with GDC 1 as implemented by Regulatory Guide 1.150, requiring ultrasonic testing of reactor vessel welds during preservice and inservice examinations.

Basis:

The Applicant has stated that it does not intend to comply with all the terms of Regulatory Guide 1.150, yet it does not indicate any alternative ways in which the requirements of the Reg. Guide will be satisfied. The Applicant's statement that it agrees with the "intentions" of Reg. Guide 1.150 does not constitute sufficient compliance with the Reg. Guide. Since the specific implementation plan for Reg. Guide 1.150 has not been submitted, NECNP reserves the right to amend this contention to challenge the sufficiency of its provisions.

I. D. 2. The Applicant has not complied with GDC 21 as implemented by Regulatory Guide 1.22, requiring periodic testing of protection system actuation functions. In particular, there are twelve safety functions which the applicant does not plan to test at power:

1. Generation of a reactor trip by tripping the main coolant pump breakers.
2. Generation of a reactor trip by tripping the turbine.
3. Generation of a reactor trip by use of the manual trip switch.
4. Generation of a reactor trip by manually actuating the safety injection system.
5. Generation of safety injection signal by use of the manual safety injection switch.
6. Generation of containment spray signal by use of the manual spray actuation switch.
7. Full closure of main steam isolation valves.
8. Full closure of feedwater isolation valves.
9. Full closure of feedwater control valves.
10. Main feedwater pump trip.
11. Closure of reactor coolant pump component cooling water isolation valves.
12. Closure of reactor coolant pump seal water return valves.

FSAR at 1.8-9. The design and operation of the facility should be changed to provide for testing at power in all of these cases.

Basis:

GDC 21, as implemented by Regulatory Guide 1.22, requires that actuation of safety functions be tested at power. Otherwise sufficient assurance cannot be provided that it will be able to function while the reactor is operating. This is a fundamental require-

ment that cannot be waived by an unsupported assertion that the probability of failure at power is too low. The design of the Seabrcok facility should be revised, if necessary, to allow testing at power for these necessary safety system actuations.

I. D. 3. The Applicant has not provided a reasonable assurance that the leakage detection system for the Seabrook reactor will operate when needed because not all of the system is to be tested during plant operation, as required by GDC 21, and as implemented by Reg. Guide 1.22. Only the airborne radioactivity detector has the capacity to be tested during power operation. FSAR at 1.8-17. The Applicant thereby also fails to satisfy GDC 30, which requires the development of adequate leakage detection systems.

Basis:

GDC 21, the Staff's Standard Review Plan and IEEE Standard 279-71 all require that safety systems be capable of being tested at power. Otherwise, there can be no reasonable assurance that the equipment will operate under the conditions in which it will be called upon to function. The Applicant has failed to justify its failure to comply with GDC 21 or to demonstrate that it will provide protections equivalent to those provided by compliance with Reg. Guide 1.22.

I.D. 4. The Applicant has not complied with the terms of Regulatory Guide 1.118, Rev. 2, requiring periodic testing of electric power and protection systems. In particular, the Applicant indicates compliance with an outdated standard, IEEE 338-1975, which has been superseded by IEEE 338-1977. Furthermore, the Applicant improperly asserts that it need not comply with IEEE 338-1975 whenever the standard states that an action "should" be taken or a requirement "should" be met. FSAR at 1.8-41. All the provisions of the IEEE standard should be treated as mandatory unless the applicant can show an alternative means of satisfying them.

Basis:

Regulatory Guide 1.118 Rev. 2 incorporates IEEE 388-1977 as the standard to be applied in determining the requirements for compliance with the Regulatory Guide. The applicant may not substitute another standard of its own choosing without demonstrating that it has provided equivalent protection. It has made no such showing.

The fact that IEEE asserts that actions or requirements "should" be met constitutes a factual basis for NECNP's contention that they must be met in order to assure the safety of the Seabrook facility. At the very least, the applicant should show that it has satisfied the requirement by some other means if it does not met the IEEE standard itself. By stating that its compliance with these provisions is discretionary, the Applicant witholds any commitment to continue to comply with them in the future. Thus, it does not provide the requisite reasonable assurance that the plant can be operated safely.

I. E. Reactor Coolant Pump Flywheel Integrity

The Applicant has not complied with GDC 4 as implemented by Reg. Guide 1.14 in that it has not met all the requirements of Reg. Guide 1.14 or provided for suitable alternative means of satisfying the requirements. In particular, the Applicant should be required to perform post-spin inspections of the flywheel, should identify the design speed of the flywheel and test it at 125% of that speed, and should specify the cross-rolling ratio. Furthermore, the flywheel should be environmentally qualified under GDC 4 because it constitutes equipment important to safety.

Basis:

GDC 4 requires that equipment important to safety be protected from missiles. In addition, it requires that equipment important to safety be able to function when called upon to mitigate the effects of an accident. The flywheel is both a potential source of damaging missiles, and a component important to safety because it provides inertia to ensure a slow decrease in coolant flow in order to prevent fuel damage as a result of a loss of power to the pump motors. According to Reg. Guide 1.14,

> Overspeed of the pump rotary assembly during a transient increases both the potential for failure and the kinetic energy of the flywheel. The safety consequences could be significant because of possible damage to the reactor coolant system, the containment or other equipment or systems important to safety.

The Applicant's FSAR is therefore inadequate because the flywheel has not been environmentally qualified; and because not all of the requirements of Reg. Guide 1.14 have been met or a reasonable alternative implemented. Under Reg. Guide 1.14, the Applicant

should be required to identify the design speed of the Seabrook flywheel, which should be at least 125% of normal speed. Applicant should be required to test the flywheel at that speed, and to perform post-spin inspections, as required by Reg. Guide 1.14. Applicant's assertion that calculations demonstrate a sufficiently low probability of flaw growth is not an adequate justification for waiving the post-spin inspection. FSAR at 1.8-6. Finally, as required by Reg. Guide 1.14, flywheel plate material should be cross-rolled at a ratio of 1 to 3. The Applicant cannot waive this requirement by relying on material selection, particularly in light of the inadequacies in vendor surveilance that have occurred at Seabrook.

I. F. Diesel Generator Qualification

According to the FSAR discussion of Regulatory Guide 1.9 at page 1.8-4, the "Load Capability Qualification" test for the Seabrook diesel generators was performed according to the requirements of IEEE 387-1977. NECNP contends that the diesel generators cannot be considered to be qualified for use at Seabrook on that basis. They must also, at a minimum, meet the requirements of IEEE 323-1974.

Basis:

The basis for this contention is the NRC Staff position set out in Regulatory Guide 1.9, which provides that the qualification testing requirements of IEEE 323-1974 should be used in section 5.4 of IEEE 387-1977. Based on the FSAR, the Applicant has failed to do so, and it has failed to demonstrate that it has provided protections equivalent to those provided by Reg. Guide 1.9.

I. G. Pressure Instrument Reliability

The Reactor Cooling System wide range pressure instruments for the Seabrook facility cannot be relied upon for accurate information, and thus may lead to inappropriate operator actions jeopardizing the cooling of the reactor. The pressure instruments therefore do not provide the requisite reasonable assurance of safe operation of the plant.

Basis:

According to IE Information Notice No. 82-11 (April 9, 1982), qualification tests on Westinghouse-manufactured RCS pressure instruments have shown "ambiguities in their accuracy which could result in inappropriate operator actions." In particular, Westinghouse told the NRC that :

> post-accident accuracy ambiguities for RCS wide range pressure instruments under certain plant accident conditions have the potential for maximum accumulated inaccuracies of ± 390 psig indication. Accordingly, the inaccuracy of RCS wide range pressure measurements could lead to pressurizer power operated relief valves being lifted prior to the termination of safety injection (SI) and to a greater number of valve challenges, thereby increasing the probability of a small break loss-of-coolant accident due to a valve failing to close. Likewise, the inaccuracy of the wide range pressure instruments could lead to the termination of SI without adequate reactor coolant subcooling. In addition, the inaccuracies could lead to premature or late tripping of the RCS pumps during the course of a small break loss-of-coolant accident.

Because the pressure instruments may provide inaccurate information leading to the exacerbation or failure to correct accident conditions, their use constitutes a threat to the public health and safety and cannot support a license for the Seabrook reactor.

I. H. Decay Heat Removal

One of the lessons of the Three Mile Island accident was that heat exchanger capacity in nuclear power plants should be expanded and improved. This is particularly true with respect to the unexpectedly lengthy period it took to cool the TMI reactor and the need to assure effective heat exchange at high pressures. However, the Applicant's proposed heat exchanger capacity is even lower than a number of older plants of comparable design. FSAR Table 1.3-1, Sheet 2. The Applicant should be required to install additional heat exchanger capacity to allow for more rapid cooldown of the facility in the event of an accident.

Basis:

The current inadequacy of decay heat removal systems has been noted as an unresolved safety issue:

> Even though it may generally be considered safe to maintain a reactor in a hot standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cool enough to perform inspection and repairs. For this reason the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs.

NUREG-0510, Identification of Unresolved Safety Issues Relating to Nuclear Power Plants (January 1979) at A-15. The issue was further recognized by the formal addition of Task A-45 to the list of unresolved safety issues as a result of the TMI accident. NUREG-0705 at A-1. The Reactor Safety Research Review Group found in 1981 that:

> A major effort should be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems, for both the reactor vessel and the containment.

Report of the Reactor Safety Research Review Group, September 1981, at II-3. The principle of expanded and improved heat removal capacity is violated by the reduction in size of the heat exchanger capacity for Seabrook.

I. I. Inadequate Provisions for Achieving Cold Shutdown

Regulatory Guide 1.139 establishes specific design requirements that address the various system functions required to achieve and maintain a safe hot standby and cold shutdown position. According to I&E Bulletin 79-01B, Supp. 3, October 24, 1980, it is the position of the NRC Staff that "the licensee must identify and environmentally qualify the equipment needed to complete one method (path) of achieving and maintaining a cold shutdown condition." NECNP contends that Seabrook does not conform to NRC requirements and constitutes a threat to the public health and safety because, as described in the FSAR, particularly at pages 1.8-52-54, it is not capable of being brought directly to cold shutdown in the event of an accident. Further, the Applicant has failed to demonstrate that all systems, structures, and components necessary to bring the facility to cold shutdown have been environmentally qualified or that they have met all of the design criteria applicable to systems, structures, and components important to safety, including but not limited to G C 3,4,15,17,18,20-25,34,35,44.

Basis:

The basis for this contention is the position of the NRC Staff as set out in I&E Bull. 79-01B, Supp.3, and Reg. Guide 1.139, and the Applicant's admission in the FSAR that it does not meet the Reg. Guide. The Applicant has provided only "systems with the capability to place and maintain the plant in a safe hot standby condition," such that a restoration of some degree of systems capability would be required to bring the plant to cold shutdown in the event of an accident. The Applicant has failed to demonstrate

that it has provided equivalent protection. In addition, this contention is based on the fact that the Applicant has failed to demonstrate compliance with the exceptions to Reg. Guide 1.139 discussed at page 1.8-53 of the FSAR, the fact that the TMI accident demonstrated the need to have the capability of achieving cold shutdown in the event of an accident, and the need to assure the environmental qualification of all systems, structures, components, and functions necessary to achieve cold shutdown. To the extent that operator actions are relied upon to achieve cold shutdown, the function is not environmentally qualified and does not meet the applicable requirements.

I. J. Sabotage

10 CFR Part 73, and particularly Sections 73.40-73.55, require the Applicant to develop and implement a plan that would effectively protect the Seabrook reactors against industrial sabotage. Regulatory Guide 1.17, Rev. 1, issued in June 1973, establishes the requirements and procedures that the NRC Staff believes would be sufficient to comply with the regulations and provide the necessary protections. NECNP contends that the Seabrook reactors are seriously vulnerable to industrial sabotage by virtue of their design and that the Applicant's security plan is inadequate to prevent actions of industrial sabotage at Seabrook that would threaten the public health and safety.

Basis:

The Applicant's FSAR itself provides a prima facie basis for this contention when it states that the Seabrook security plan "meets the intent of Regulatory Guide 1.17, Rev. 1." In this regard, NECNP has obtained the assistance of Robert Pollard, fomerly of the NRC Staff, now with the Union of Concerned Scientists, who has found in his many years of reviewing FSARs that a statement that an aspect of a nuclear plant meets the "intent" of the Regulatory Guide constitutes an admission that the plant cannot meet actual specifics of the Reg. Guide. It is also instructive to compare the FSAR language on this issue with other FSAR statements in which the Applicant asserts that it is in full compliance with Regulatory Guides. While Regulatory Guides do not constitute firm requirements, they do indicate what the NRC Staff believes to be one manner of complying with the regulations, and they provide a basis for judging the adequacy of other methods by comparing them to

the Staff position. Failure to comply with a Regulatory Guide in this case constitutes a factual basis for this contention.

In addition, the history of industrial sabotage in recent years at the Surry, Midland, Beaver Valley, and Indian Point Unit 2 reactors demonstrates that security plans that were presumably developed and approved as complying with or meeting the intent of Regulatory Guide 1.17 failed to provide the required protection.

It is not possible to provide further basis or specificity since the Applicant's security plan has not been made availabe for review. However, Mr. Pollard has confirmed on the basis of his expertise the vulnerbility of the Seabrook reactors to industrial sabotage and the fact that previous incidents of industrial sabotage demonstrate that the same or worse could occur at Seabrook. Mr. Pollard has considerable experience in addressing the issue of industrial sabotage both while he was on the NRC Staff and since he left. His expertise is in reactor design as it relates to sabotage, rather than in the technical aspects of security planning itself, although his expertise is essential to an adequate plan. He is prepared to examine the Seabrook design in connection with the security plan in order to identify the improvements that are required to assure safety. An example of an aspect of the design that is vulnerable to industrial sabotage is any valve that is intended to remain open in the event of an accident, but that could be manually closed without detection prior to an accident, with the result that recovery from the accident would be hindered or prevented. This occurred at Beaver Valley. NECNP would also attempt to obtain the assistance of an expert in security plans to evaluate whether the improvements identified by Mr. Pollard can be accomplished, and if so, how.

NECNP and Mr. Pollard are prepared to provide any assurances against disclosure that may be required to protect the integrity of the security plan and the public health and safety. We will also provide additional specifity and basis after we have had an opportunity to review the security plan.

I. K. Instrumentation for Monitoring Accidents

The Applicant has not satisfied GDC 13, 19, and 64, as implemented by Reg. Guide 1.97. The General Design Criteria and the Regulatory Guide relate to the instrumentation required to monitor plant conditions both during and after an accident. The instrumentation should be environmentally qualified.

Basis:

The basis for this contention is that the Applicant has given no information to show satisfaction of the requirements. The Applicant instead indicates that it is still in the process of selecting the Post-Accident Monitoring instrumentation for the facility. FSAR 1.8-36. NECNP reserves the right to amend this contention to challenge the adequacy of the PAM if and when it is submitted to the NRC by the Applicant.

I. L. PORV Flow Detection Monitoring System Inadequate

According to Table 1.3-2 of the FSAR, the Applicant has added a three channel acoustic accelerometer system to comply with the requirement to detect flow from PORV's and safety valves. NECNP contends that this monitoring system is inadequate to protect the public health and safety and fails to comply with the minimum requirements of the NRC. Further, whatever monitoring system is provided must be safety-grade and environmentally qualified.

Basis:

One of the major lessons learned from the accident at Three Mile Island is that the instrumentation and monitoring systems used to measure the various conditions in the reactor during an accident are both crucial to reactor safety and seriously inadequate in many respects. In particular, the NRC Staff requires the use of a positive, direct indication of valve position rather than the indirect measurement previously used. This is based on both the TMI accident experience and on IEEE 279, which "requires that, to the extent feasible and practical, protection system input shall be derived from signals that are direct measures of the desired variable." NUREG-0578, p. A-9.

Contrary to these lessons and requirements, the Applicant is relying upon an indirect measure of protection system input by measuring noise rather than measuring the actual flow from the power operated relief valves and the safety valves. Safe reactor operation requires that the acoustic accelerometer system be replaced with a monitoring system that directly measures the flows.

I. M. Fire Protection

The Applicant's fire protection system does not satisfy GDC 3 as implemented by Reg. Guide 1.120 and the Commission's decision in CLI-80-21. In particular, the Applicant indicates that it does not comply with all aspects of the Standard Review Plan, Section 9.5-1, but does not specify what those items are or any alternative means of satisfying the requirements. Furthermore, the information submitted to the NRC in 1977 is outdated and should be revised and made available as part of the FSAR.

Basis:

The Commission's decision in CLI-80-21 requires that Applicants whose construction permit applications were docketed before July 1, 1976, demonstrate compliance with Appendix A to BTP 9.5-1 and the requirements set forth in the proposed rule, which was finalized at 45 Fed. Reg. 76602.

The Applicant indicates that it does not comply with all requirements of the Standard Review Plan at 9.5-1. FSAR at 1.8-43. This information, which was submitted in 1977, is outdated and should be revised to reflect more recent developments, including changes in SRP 9.5. The Applicant should be required to specify which items are not complied with and to specify alternative means of satisfying the requirements. NECNP reserves the right to comment on the sufficency of the Applicant's fire protection plan when this information is made available.

I. N. Solid Waste Disposal System

The Applicant has not submitted a complete solid waste management plan for radioactive waste as required by GDC 60 and implemented by Reg. Guide 1.143.

Basis:

General Design Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational occurrences. The Applicant should be required to complete its solid waste management plan for radioactive waste in satisfaction of this requirement. NECNP reserves the right to challenge the sufficiency of the plan if and when it is submitted in final form.

I. O. Emergency Feedwater

1. The emergency feedwater system is inadequate in that a single failure in the common discharge header, in conjunction with delayed operator action or no action to correct it, would result in a loss of feedwater to all the steam generators. The Seabrook design must be revised to provide an emergency feedwater system that is single failure-proof with respect to a rupture of the high-energy piping in the discharge header. Even if the common discharge header is not considered to be covered by the single failure criterion, the Applicant has not adequately considered the factors necessary to protect against passive system failure.

Basis:

The emergency feedwater system design for the Seabrook facility provides one common discharge header for all the steam generators. This system is placed under stress and pressure when the emergency feedwater system is activated. In the event of a rupture in the common discharge header, feedwater supply to all the generators would be jeopardized. Such a rupture should meet the Single Failure Criterion of Appendix A to 10 CFR Part 50.

Even where systems are not specifically required to meet the Single Failure Criterion, the Applicant must consider the possibility of a single failure. Footnote 2 to Appendix A states that "the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development." However, this statement

> "does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary requirements. These matters include:

(i) Consideration of the need to design against single failures of passive components in fluid systems important to safety.

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of the protection system and reactivity control systems.

36 Fed. Reg. 3255 (February 20, 1971), preamble to 10 CFR Part 50 Appendix A.

At Seabrook, in the absence of prompt operator action to correct a loss of feedwater, all of the steam generators would be threatened by loss of coolant. Reliance on such operator action is unacceptable. To satisfy the Single Failure Criterion and the considerations listed in the preamble to Appendix A, the Applicant should redesign the facility to provide redundant feedwater capacity or institute automatic initiation of the emergency feedwater system.

I.O.2. The emergency feedwater system is inadequate in that a break in the common discharge header between Valve 65 and Valve 125 (see FSAR, Figure 6.8-1), coupled with a loss of offsite power, would result in a loss of feedwater to all steam generators. The Seabrook design must be revised to provide an emergency feedwater system that complies with GDC 17 with respect to the high-energy piping in the discharge header.

Basis:

The applicant has failed to meet its own criteria with regard to loss of power and single failure in high-pressure systems. In Section 6.8.3, the FSAR states: "The system has been designed to provide the required flow following a single active failure coupled with a passive failure in the high or moderate energy piping and a loss of offsite power." However, the FSAR improperly fails to include the common discharge header in the class of "high or moderate energy piping" because it is "not pressurized during normal plant operation." In analyzing the adequacy of safety functions, normal operating conditions cannot be the basis for classification of the type of equipment which must be considered in the analysis, because normal operating conditions are irrelevant to how the equipment will behave under accident conditions. The classification of the common discharge header as unpressurized is invalid because the header will be pressurized when it is called upon to function. The applicant therefore has no basis for claiming that the system is designed to provide needed feedwater flow in the event of a single active failure and a moderate or high energy piping failure coupled with a loss of offsite power.

I. P. Human Engineering

According to Table 1.3-2, Sheet 8 of the FSAR, the Applicant has added a 0-2300° F multipoint recorder on the back of the main control panel. Its purpose is to record temperature at four core locations. NECNP contends that the location of this recorder on the back of the control panel constitutes poor human engineering that would detract from the operator's ability to take prompt, correct actions in the event of an accident.

Basis:

This contention is based on the fact that information that may become of major interest to the operator will be available only on the back of the control panel. The operator will be required to leave his station and divert his attention from on-going events in order to determine the temperature in the core as stated on the multipoint recorder. The information should be readily available such that the operator need not move to the back of the control panel.

I. Q. Systems Interaction

The Applicant has not demonstrated that it has adequately evaluated systems interaction to confirm that Seabrook has been designed against all potential undesirable interactions between and among systems in order to meet the requirements of 10 CFR Part 50, Appendix A.

Basis:

One of the so-called "generic unresolved safety issues" is the interaction between non-safety and safety systems, which creates demands on the safety systems that exceed the latter's design basis. This problem is listed and described as A-17 in NUREG-0510.

The NRC, in investigating an accident at the Zion plant on July 12, 1977, discovered a design defect in the Westinghouse plant in that Westinghouse designs provide for a large number of, and different types of, interactions between control systems and safety systems. The Applicant has not demonstrated that these design deficiencies have been modified or remedied at Seabrook.

On October 12, 1979 the ACRS recommended with respect to the Indian Point plant and all light water reactors that the NRC study systems interactions by investigating, inter alia, sub-system failures within interconnected electrical

or mechanical complexes and potential interactions between nonconnected systems. The latter investigation may require in-situ examination of the plant.

On March 9, 1982, the ACRS again recommended to the NRC chairman that a walk-through systems interaction study be developed for all light water reactors to detect obvious interactions.

The Standard Review Plan states that a systematic methodology is necessary for evaluation of systems interactions Sandia Laboratories, under contract to the NRC, has stated that the methodology used by the NRC staff is inadequate to identify all interactions important to safety. In one study of systems interaction for earthquakes at Diablo Canyon, about 600 previously undetected systems interactions were found. The Applicant was required to make plant modifications.

The Report of the Reactor Safety Research Review Group, issued September 1981, recommended intensified research to define better the role of plant control systems in light water reactors. The recommendation was prompted by severe transients initiated by control systems in recent years in light water reactors such as Seabrook.

According to indications in the Applicant's letter to the Commission dated January 8, 1982, it has not satisfied the Staff's questions about the safety consequences of interactions between control systems and safety systems at Seabrook.

The recent "state of the art" review, conducted by Brookhaven National Laboratory, Battelle Columbus Laboratories and Livermore Laboratories concluded that no single method currently exists to perform an adequate review of adverse systems interactions. NUREG/CR-1901, Review and Evaluation of System Interactions Methods, A.G. Buslick et al, Brookhaven National Laboratory, January, 1981; NUREG-1896, Review of Systems Interaction Methodologies, prepared by P. Cybulskis et al., Battelle Columbus Laboratories, January, 1981; NUREG/CR-1859, Systems Interaction, State-of-the-Art, Review and Methods Evaluation, J.J. Lim et al., Lawrence Livermore Laboratory, January 1981.

The Applicant indicates that it meets only the current Regulatory Guides, which the Commission itself believes to be inadequate to deal with the safety effects of systems interaction. Therefore the Applicant's analyses of systems interactions, contained in Chapter 15 of the FSAR, are inadequate.

Furthermore the Appeal Board in Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978) ruled that as a requirement for the issuance of an operating license the record must demonstrate that each applicable generic safety issue has been resolved for the particular reactor, or demonstrate the existence of measures employed at the plant to compensate for the

lack of a solution to the problem. The Applicant has failed to demonstrate that it has resolved this generic safety problem or that it has devised compensating measures.

I. R. Hydrogen Control System

The design of the hydrogen control system at Seabrook is inadequate to protect the public safety in that it would protect against the hydrogen produced by a metal water reaction involving only 1.5% of the fuel cladding. FSAR 1.8-3.

Basis:

The accident at Three Mile Island demonstrated that as much as 50 percent of the zirconium cladding in the TMI core reacted chemically to produce hydrogen, an amount greatly in excess of the design assumptions of 10 CFR 50.44. The NRC, in its Proposed Rule of December 18, 1981, "Interim Requirements Related to Hydrogen Control," stated that two years after the effective date of the rule all operators would be required "to perform and submit . . . analyses to assure that during a degraded core accident containment structural integrity will be maintained. . . ." The Commission assumed that a degraded core accident would involve a 75% metal-water reaction, with the resulting massive hydrogen release to the containment.

Furthermore, in the Commission's current Proposed Policy Statement related to Safety Goals for Nuclear Power Plants (Feb. 11, 1982) the Commission explicitly recognized the importance of mitigating the consequences of a core-melt accident through assuring the integrity of the containment in the event of a hydrogen release.

The Commission in Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1), CLI-80-16, 11 NRC 674 (1980), appeared to require that an intervenor contesting the adequacy of a plant's hydrogen control system hypothesize a credible accident scenario in which the hydrogen generated would cause release of off-site radiation levels in excess of 10 CFR Part 100 limits, and denied, in that case, the intervenor's request to waive the clearly inadequate design basis contained in 10 CFR 50.44. Yet in a more recent Commission decision, the Commission itself recognized the inadequacy of 50.44 and required an applicant to install and use an igniter hydrogen mitigation system to mitigate the possible release of hydrogen in excess of the assumptions of 50.44, and to continue a research program on hydrogen control measures and the effects of hydrogen burns on safety functions. Duke Power Company (McGuire, Units 1 and 2) CLI-81-15, 14 NRC 1 (1981).

Another licensing board admitted a similar contention on the inadequacy of a plant's hydrogen control system on the basis that "Commission utterances, proposed and tentative though they may be, [appear] to be inconsistent with the TMI decision on which we relied. The Commission now appears to be of the view that the assumptions of 50.44 are unrealistic and that some additional [illegible] may need to

be taken." Cleveland Electric Illuminating Corp. (Perry Nuclear Power Plant, Units 1 and 2), Nos. 50-440, 50-441, March 3, 1982, slip op. at 8.

The Applicant has stated that Seabrook is designed on the assumption that no more than 1.5 percent of the cladding will react to produce hydrogen, FSAR at 1.8-3, and that it has met the requirements of Regulatory Guide 1.7 only with regard to the Westinghouse-supplied components of its hydrogen control system. Thus the Applicant has failed to demonstrate that Seabrook's hydrogen control system is adequate to control the amount of hydrogen created in loss-of-coolant accidents in light of the experience from the TMI accident.

I. S. Loose Parts Detection System

The Applicant has not yet designed or developed a loose-parts detection system for the reactor's primary system, and therefore does not satisfy Criteria 1 and 13 of Appendix A to 10 CFR Part 50, 10 CFR 50.36, or 10 CFR 20.1(c), as implemented by Regulatory Guide 1.133, and does not provide an adequate alternative to satisfy the requirements.

Basis:

Regulatory Guide 1.133 describes an acceptable method to implement NRC requirements with respect to detecting potentially safety-related loose parts in light water cooled reactors during normal operation. By complying with Regulatory Guide 1.133 an Applicant will satisfy the NRC staff that Criteria 1 and 13 of 10 CFR Part 50, section 50.36 of 10 CFR Part 50, and Paragraph 20.1(c) of 10 CFR Part 20 have been met.

The Applicant has not designed or developed a loose parts detection system, and thus has not met the NRC's design, instrumentation and technical specifications requirements for safe operation of Seabrook. In addition, the Applicant has not demonstrated, with respect to the handling of loose parts, that it can meet Part 20 criteria which

mandate that licensees make every reasonable effort to maintain exposures to radiation as far below Part 20 limits as reasonably achievable.

Moreover, the Applicant has not demonstrated that it has designed or developed an adequate alternative method of implementing the above requirements.

After the Ginna accident, a TV-optics examination identified foreign objects and tube fragments inside the B-steam generator. An examination of the A-steam generator also revealed the existence of some small foreign objects. The NRC report states that neither the utility nor the NRC has conclusively determined the cause of the tube rupture, which raises the possibility that these loose parts, probably left from recent repair work on the steam generators, caused the accident. NUREG-0909.

Therefore, the Applicant should, at a minimum, demonstrate that its loose parts detection s tem for Seabrook meets the minimum requirements of Regulatory Guide 1.133, especially since the history of Westinghouse steam generators has shown the need for frequent repair work and thus the likelihood of loose parts being left near the generators.

I. T. Steam Generators

The Applicant has not demonstrated that the Seabrook steam generators are capable of resisting degradation or that the new Model F Westinghouse generators have been designed to solve the degradation problem and maintain their integrity during normal operation and during a credible accident scenario such as the accident which occurred at Ginna on January 25, 1982.

Basis:

The Report of the Reactor Safety Research Review Group, issued September 1981, advised the NRC that so-called steam generator tube degradation is a problem which "has not been considered sufficiently using recent accident analysis codes" in order to estimate "the consequences of a transient or some other failure that might lead in turn to the failure of a significant number of tubes. Such failures could lead to the degradation of ECCS function."

Among the problems with Westinghouse steam generators since June, 1977, are the following:

1) Tube decay and support plate cracking (related to denting) at Indian Point Unit 2;

2) Tube denting and discovery of support plate cracking at San Onofre Unit 1;

3) Tube denting at Surry Units 1 and 2, and Turkey Pont Units 3 and 4.

The rupture of a large-scale generator tube in the Ginna accident combined with the failure of a pressurizer PORV created a situation in which operators had to choose between possible exposure of the core and release of radioactive steam.

Westinghouse claims, but has not actually demonstrated, that its improved Model F design has features that reduce the potential for denting and that it has changed its chemistry program to ensure tube integrity against thinning. However, the Ginna accident demonstrated that serious problems with Westinghouse steam generator tubes continue, and that the Applicant should be allowed to operate with such tubing only after extensive testing and analysis of the steam generator tubes, and an affirmative demonstration that the probability of their thinning, denting or rupture causing an ECCS steam binding emergency or other serious event is acceptably low.

I. U. Turbine Missiles

The Applicant has not demonstrated that it meets GDC 4 of Appendix A to 10 CFR 50, as implemented by Regulatory Guide 1.115, which requires that structures, systems, and components important to safety be protected against the efeects of turbine missiles whose launching might occur as a result of equipment failure. Nor has the Applicant demonstrated that it has designed an adequate alternative method to meet GDC 4.

Basis:

The Applicant states that it does not comply with Regulatory Guide 1.115, Rev. 1, and that the probability of damage due to low trajectory missiles is greater than 10^{-3}. The Applicant has failed to demonstrate in subsection 3.5.1.3 of the FSAR that the Seabrook plant has an acceptable alternative method to meet GDC 4, or that it has met Regulatory Guide 1.115 which provides an acceptable method to comply with GDC 4.

As can be seem from Figure 3.5.1 of the FSAR, certain low-trajectory missiles resulting from turbine failure could severely harm the containment of one or both of the two Seabrook plants.

The Applicant, furthermore, has admitted that it cannot demonstrate, as the NRC requires, that the probability of damage due to such low-trajectory missiles is lower than 10^{-3}.

In-service inspection of steam generators has been demonstrated historically to be inadequate to prevent their degradation and resulting accidents due to this degradation. The Applicant has not stated that it fully meets all requirements of Regulatory Guide 1.83, which sets forth one acceptable method of ensuring that in-service inspection of steam generator tubes complies with General Design Criteria 14, 15, 31, and 32 of Appendix A to 10 CFR Part 50.

Therefore, the Applicant must redesign the Seabrook plant to ensure full compliance with GDC 4.

I. V. In-Service Inspection of Steam Generator Tubes

The Applicant has not demonstrated that it meets General Design Criteria 14, 15, 31 and 32 of Appendix A to 10 CFR Part 50, as implemented by Regulatory Guide 1.83, in order adequately to reduce the probability and consequences of steam generator tube failures through periodic in-service inspection for early detection of defects and deterioration. Nor has the Applicant developed an adequate alternative program for in-service inspection of steam generator tubes.

Basis:

The Applicant has not demonstrated, in subsection 5.4.2.5 of the FSAR, that its program for in-service inspection of steam generator tubes meets Regulatory Guide 1.83, which provides one acceptable method of complying with the requirements of General Design Criteria 14, 15, 31 and 32 of Appendix A to 10 CFR Part 50. Nor has the Applicant designed an adequate alternative method.

The Applicant says merely that the inspection program "will be performed in accordance with the requirements of Regulatory Guide 1.83, Rev. 1," and that the steam generators are designed to permit access to tubes for inspection or plugging repairs. See FSAR 1.8-31 and 5.4-19.

Given the long history of serious problems with Westinghouse steam generators, and the recent accident at

Ginna caused by a rupture in a steam generator tube, it is imperative that the Applicant demonstrate that its inspection program is adequate to provide some assurance that problems will be found and remedied in steam generator tubes prior to another Ginna-type accident. Even though Ginna and a number of other plants have satisfied the NRC staff of the adequacy of their in-service inspection and repair programs, the continuing problems with Westinghouse steam generators and the serious accident at Ginna demonstrate that the requirements of Regulatory Guide 1.83 are not sufficient to reduce the probability and consequences of steam generator tube failures through periodic inspection for early detection of defects and deterioration.

The NRC found after the Ginna accident that the ruptured tube had been inspected as late as April 1981, and found not to have experienced any significant degradation. The Ginna in-service inspection program also appeared to comply with the requirements of Regulatory Guide 1.83. NUREG-0909. Accordingly, the April 11, 1981, inspection failed to accomplish its most important goal--prevention of an accident that might result from a defective tube.

I. W. Seismic Qualification of Electrical Equipment

The Applicant has not demonstrated that it has adequately assured the seismic qualification of electrical equipment at Seabrook, as required by Criterion III, "Design Control" of Appendix B to 10 CFR Part 50 and implemented by Regulatory Guide 1.100, Rev. 1. The Applicant also has not shown that it has developed an adequate alternative method of seismically qualifying electrical equipment.

Basis:

According to FSAR 1.8-36, the Applicant has not demonstrated that all NSSS safety-related electrical equipment or BOP electric equipment has been seismically qualified to meet all requirements of Regulatory Guide 1.100, Rev. 1. However, in a letter to the NRC the Applicant states that qualification of electrical equipment and instrumentation complies with the guidelines of Regulatory Guide 1.100.

Because of the conflict between the statements of the Applicant in FSAR 1.8-36, and those statements made in its January 8, 1982 letter to the NRC, the Applicant must demonstrate that its method of seismically qualifying electrical equipment at Seabrook fully complies with Criterion III of Appendix B. to 10 CFR Part 50.

Seismic Qualification of Equipment is an Unresolved Safety Issue listed as A-46 of NUREG-0705. The Applicant has not demonstrated that this unresolved safety issue has been resolved for Seabrook or that measures exist and are employed at the plant to compensate for the lack of a solution to this unresolved safety issue, as required under the Appeal Board decision in Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978).

II. QUALITY ASSURANCE CONTENTIONS

A. Design and Construction

1. General Design Criterion 1 of Appendix A to 10 CFR Part 50 requires the establishment and implementation of a quality assurance program. This and all General Design Criteria cover all aspects of the facility that are "important to safety." NECNP contends that the Seabrook Quality Assurance Program for design and construction has been too narrow in scope, applying only to items considered to be "safety-related," rather than to the broader category of aspects that are "important-to-safety." Accordingly, the Applicant has failed to comply with GDC 1 to Appendix A.

Basis:

According to page 105 of the Regulatory Agenda issued by the NRC on January 31, 1982, the original intent of the Commission in issuing Appendix B to 10 CFR Part 50 was to require that the Quality Assurance programs required by that Appendix cover "the full range of safety matters," rather than some subset that is considered to be "safety-related." The Regulatory Agenda cites examples of structures, systems, and components for which Appendix B criteria have not been fully implemented, including in-core instrumentation, reactor coolant pump motors, reactor coolant pump power cables, and radioactive waste system pumps, valves, and storage tanks.

Section 17.1.1.2(b) of the FSAR describes the aspects of the Seabrook facility covered by the Quality Assurance Program as the "safety-related structures, systems, and components" listed in Tables 17.1-1, 17.1-2, and 17.1-3." None of the examples cited by the Commission as important to safety is covered by the Seabrook Quality Assurance Program. Another example of aspects of the facility that are important to safety, but are not included in the FSAR tables is all equipment that removes heat from the steam generators during shutdown. Such requirement is essential to assure adequate decay heat removal as required by GDC 24.

II. A. 2. Under 10 CFR Part 50, Appendix B, the Applicant is required to develop and implement an effective Quality Assurance Program the provides the following assurances: (1) that the design of the Seabrook plant complies with all applicable regulations and requirements and assures the protection of the public health and safety, and (2) that all aspects of the construction of the facility are carried out in a quality manner in conformance with the design and the applicable requirements. NECNP contends that the Applicant has failed to meet the requirements of Appendix B with respect to either the design or construction of Seabrook. The Quality Assurance/Quality Control Program for Seabrook has been subject to pervasive inadequacies in all areas such that there is no assurance that the plant has been designed or constructed in accordance with applicable requirements and consistent with the protection of the public health and safety. It may not be possible to determine whether the plant has been designed and constructed to assure safety. At a minimum, a complete independent audit of all design and quality assurance documentation, a complete physical reinspection of all aspects of the plant, and a thorough conservative engineering analysis of all aspects of the plant that cannot now be inspected are required to provide a reasonable assurance of protecting the public health and safety.

Basis:

This contention is based on the performance of the Applicant as reflected in reports of the NRC Office of Inspection and Enforcement, the inadequacies in nuclear industry quality assurance demonstrated by investigations or unforseen incidents at a number of plants, including Diablo Canyon, Zimmer, Midland, South Texas, Brown's Ferry, North Anna, Davis Besse, and Rancho Seco, and on the fact that the NRC itself, in the person of Chairman Paladino, recognizes that Quality Assurance Programs to date have not been adequate. It is also based on the fact that the findings reflected in NRC I&E Reports are not the result of a thoroughgoing, systematic evaluation of all aspects of the plant, and reflect only a small percentage of the failures and violations that exist at a nuclear facility. See, e.g., Inskeep, G. W., "The Cause and Effect at Three Mile Island," Quality, May 1980, pp. 22-26.

Specifically, with respect to Seabrook, I&E reports reflect the following inadequacies or violations:

1. Acceptance of deficient conditions through apparent oversight or incompetence of inspectors. I&E Report Nos. 79-05, 79-07, 79-10, 80-06, 80-10, 80-01, 81-09, 81-12, 80-13, 82-1.*/ Appendix B, Criteria II, V, X, XIV.

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*/All I&E Reports will be identified by reference to the report number for Unit 1, Docket No. 50-43, except as otherwise noted.

2. Acceptance of deficient conditions as a result of inadequate or nonexistent Quality Assurance procedures. I&E Report Nos. 80-06, 80-04, 80-11, 81-01, 81-02, 81-03, 81-05, 81-07, 79-07, 79-06. Appendix B, Criteria II, V, XIV.

3. Failure to perform required inspections. I&E Report Nos. 79-06, 80-03. Appendix B, Criteria V, X.

4. Falsification of inspection record to show inspection was properly performed when it was not. I&E Report No. 79-06. Appendix B, Criteria II, X.

5. Failure to prevent deficiencies in pipe supports, pipe welds, and piping and tubing generally. I&E Report Nos. 80-06, 80-10, 81-03, 81-05, 81-14, 79-06. Appendix B, Criterion V.

6. Failure to determine the root causes of deficiencies or to assure that corrective acctions are taken to prevent deficiencies from recurring. I&E Report Nos. 79-06, 79-09, 80-03, 80-11, 80-12, 81-03. Appendix B, Criterion XVI.

7. Failure to assure proper design. I&E Report Nos. 81-14, 81-05. Reports pursuant to 10 CFR 50.55(e),

dated 10-27-78, 12-6-79 (three reports), 12-1-80, 7-17-81, 1-15-81, 2-23-81, 6-18-81, 8-25-81. Appendix B, Criteria III, V.

8. Failure to assure proper repairs. I&E Report Nos. 79-07, 80-04, 80-11, 80-12. Appendix B, Criteria V, IX, X.

9. Failure to assure deficiencies are not caused by poor contractor interface. I&E Report Nos. 80-11, 80-12, 81-12, 82-01. Appendix B, Criterion V.

10. Failure to assure the procurement of proper materials and failure to assure that procured items comply with all requirements. I&E Report Nos. 81-09, 81-12. Appendix B, Criteria V, VII, XV.

11. Failure to assure proper document control such that required changes are not made, and incorrect procedures and specifications are used. I&E Report Nos. 79-06, 80-03, 80-04, 80-11. Report pursuant to 10 CFR 50.55(e), dated 12-6-79. Appendix B, Criteria II, III, V, VI.

12. Pervasive deficiencies in welding and weld repairs. I&E Report Nos. 79-06, 79-07, 79-10, 80-03, 80-11, 80-10, 81-01, 81-03, 81-05, 81-09, 80-04, 80-11, 80-12. NRC Stop Work Order in letter dated 12-22-80. Appendix B, Criteria V, IX, X.

13. Inadequate audit program and inadequate commitment to and understanding of Quality Assurance. I&E Report Nos. 79-08, 78-06, 80-05, 81-12, 80-09, 78-16. Appendix B, Criteria I, II, XIII, XVIII.

In each of the areas cited above, it is clearly necessary to undertake a systematic analysis to determine the full extent of the deficiencies or inadequacies and the proper remedial actions. However, this contention is not limited to those areas. Rather, particularly by virtue of the broad range covered by these items, they demonstrate that the entire Seabrook Quality Assurance Program for design and construction is deficient.

II. B. Operations

Appendix B to 10 CFR Part 50 establishes the requirements that must be met by the Applicant in developing and implementing a Quality Assurance Program for operation of the Seabrook facility. 10 CFR 50.34(b)(6)(ii) requires that the FSAR include "a discussion of how the applicable requirements of Appendix B will be satisfied." NECNP contends that the Seabrook Quality Assurance Program for operations does not comply with either Appendix B or 10 CFR 50.34(b)(6)(ii) in the following areas:

1. The FSAR fails to address each of the criteria in Appendix B in sufficient detail to enable an independent reviewer to determine whether and how all of the requirements of Appendix B and the guidance in all applicable regulatory guides will be satisfied.
2. The Quality Assurance Program for operations extends only to matters considered to be "safety-related," and not to all structures, systems, and components "important to safety." Examples are discussed in Contention II. A. 1.
3. The Quality Assurance organization does not have the independence required by Appendix B, Criterion 1.

4. The Quality Assurance Program for operations as described in the FSAR does not demonstrate how the Applicant will assure that replacement materials and replacement parts incorporated into structures, systems, or components important to safety will be equivalent to the original equipment, installed in accordance with proper procedures and requirements, and otherwise adequate to protect the public health and safety. Similarly, the Quality Assurance program does not assure or demonstrate how repaired or reworked structures, systems, or components will be adequately inspected and tested during and after the repair or rework and documented in "as-built" drawings.

5. The Quality Assurance Program for operations as described in the FSAR fails to assure the presence on the operating staff of an adequate number of qualified QA/QC personnel, particularly during off-shifts.

Basis:

1. The basis for point 1 above is the language of Section 17.2 of the FSAR, which includes only a very general discussion of the Quality Assurance Program, with scattered references to procedures,

but does not provide the detail necessary to determine how the program will be implemented. Examples are the discussion of design control at Section 17.2.3 and document control at Section 17.2.6. Although both of these are areas in which the NRC found deficiencies during I&E inspections, the discussion in the FSAR is not sufficient to allow a determination of whether design and document control will actually be accomplished successfully. The same point applies throughout.

2. The basis for point 2 is the comparison between the scope of the Seabrook Quality Assurance Program for operations and the requirements of GDC 1 of Appendix A.

3. The basis for point 3 is the fact that the Nuclear Quality Manager reports to the Vice President - production on an equal basis with the Nuclear Production Superintendent, rather than to the Executive Vice President - Engineering and Production. Since the Vice President - Production is directly responsible for maximizing the amount of power produced by Seabrook, the quality assurance organization must report to a separate-but-equal level or a higher level in order to assure its independence and freedom of operation.

4. The basis for point 4 is the fact that the FSAR contains no discussion whatsoever of Quality Assurance for maintenance, repairs, or rework, all of which will occur during the life of the plant.

5. The basis for point 5 is the absence of any discussion in the FSAR of minimum staffing levels or any indication that sufficient Quality Assurance staffing will be assured at all times.

III. EMERGENCY PLANNING CONTENTION

NRC regulations require the license Applicant to submit with its FSAR a complete emergency plan before a license may be issued. 10 CFR 50.34(b)(6)(v). The plan must be "adequate and capable of being implemented," providing a "reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." 50.54(a)(1),(2). The emergency plan submitted by the Applicant for the Seabrook facility license is seriously deficient in a number of respects listed below, and fails to provide all the information required by Appendix E to Part 50. In its present form, the plan is incapable of being implemented or providing any assurance that in the event of an emergency adequate measures can and will be taken, and therefore it cannot be accepted as fulfillment of a licensing requirement under 10 CFR 50.47.

Specification and Basis:

1. The emergency plan does not contain an adequate emergency classification and action level scheme, as required by 10 CFR 50.47(b)(4) and NUREG-0654. No justification is given for the classification of various system failures as unusual events, alerts, site area emergencies, or general emergencies. In general, the classification scheme minimizes the potential significance of transients; for example, "failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure" is classified merely as an unusual event, in spite of the fact that the accident at Three Mile Island Unit 2 was caused in part by just such a failure. Although the

Applicant's classification scheme generally follows the scheme outlined in NUREG-0654, NUREG-0654 is not dispositive on this question. The Applicant's judgment on its classification scheme should include consideration of specific plant circumstances, such as the anticipated time lag for evacuation due to local problems. In its present form, the Applicant's classification scheme provides no assurance that the Seabrook onsite and offsite emergency response apparatus and personnel can be brought to an adequate state of readiness quickly enough to respond to an accident.

2. The emergency plan fails to identify emergency action levels or classify them according to the required responses. The symptoms of transients must be identified, monitored and responded to. The emergency plan provides no systematic means for analyzing these indicators and responding quickly to them. Guidelines should be provided for choosing appropriate responses.

3. The emergency plan does not demonstrate the applicant's ability to respond to failures at both units of the Seabrook reactor, or a failure at one unit which affects the other's capacity to operate safely. A number of factors, such as earthquake, severe storm, loss of offsite power, or degraded grid voltage, could simultaneously impair the operation of both units. There is no showing that the Applicant will have the requisite technical analysis capability or adequate emergency response personnel onsite or available within an adequate period of time in the event of an emergency affecting both units.

4. Appendix E to Part 50 requires that employees be trained for familiarity with their specific emergency response duties.

However, the emergency plan does not provide for the training of unit shift superviors to enable them to deal with special problems involved in emergencies, including making choices among alternative responses under stress.

5. The emergency plan does not indicate that local conditions were taken into account in establishing the Plume Exposure Emergency Planning Zone (EPZ), as required by 10 CFR 50.54(s)(1). As provided in that section, the 10-mile radius is only a recommendation, and constitutes the minimal bounds for an EPZ. NECNP contends that, as established, the Plume EPZ is inadequate to protect the public health and safety in the Seabrook area, considering such local factors as meterology, restricted access routes, evacuee directional bias, evacuation shadow, and the seasonal congestion of the area with summer tourists.*/

a) Meteorology: As shown by studies done for the state of California Office of Environmental Services, the frequency and pesistence of winds has a great bearing on the probability of radioactive emissions being carried in any given direction. The shape of the EPZ may well be not perfectly round but elliptical due to these factors. Furthermore, humidity, barometric pressure, and frequency of inversions may play a significant role in the dispersion of radioactivity throughout the area. The Applicant should be required to perform a complete analysis of the ways in which meteorology affects the size and shape of the EPZ.

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*/Not all of the factors discussed in this contention are mentioned specifically as local factors, but the regulation is not intended to be limited to the listed factors, as indicated by the words "such as" preceding the examples of local factors.

b) Access routes and directional biases of potential evacuees: The Seabrook area both within and beyond the 10-mile EPZ, is characterized by very restricted access routes, especially along the seashore where there is very limited access from Route 1A to Route 1 or the interstate. See FSAR Figure 4.2. These limitations should be considered in establishing the EPZ. Furthermore, studies at Three Mile Island have revealed directional biases which have a potentially significant bearing on the size and shape of the Plume EPZ. The study has suggested that three factors affect the direction that evacuees will take:

(i) A strong directional bias in favor of upwind destinations from the reactor

(ii) A "psychological attraction" to mountainous areas in time of danger

(iii) A reluctance to select destinations in more densely populated urban areas.

Ziegler, Brunn, and Johnson, "Evacuation from a Nuclear Technological Disaster", Geographical Review, Vo. 71 No. 1, January 1981, at 9. All of these factors throw off calculations based on predesignated evacuation routes, and should be specifically considered as independent variables in the determination of the proper EPZ. The information concerning evacuation preferences should be obtained directly by interviewing local residents and transients regarding their likely behavior during an accident.

c) Population Characteristics: During the summer, beaches both within and beyond the 10 mile radius are congested with people and the coastal road clogged with slow-moving traffic. This condition would be aggravated if, in an emergency, everyone tried to leave the area at once. The ETAs aren't based on a realistic model of how the evacuation network will be loaded and therefore underestimate the time it would take to evacuate the area. See

Science Applications, Inc., "A Study of Postulated Accidents at California Nuclear Plants", prepared for the State of California Office of Energy Services, PA101380-3811J p. 8-17. The difficulty of moving traffic expediently and sheltering large numbers of people on the beach in the summer months should be considered in establishing the size and shape of the EPZ.

d) Evacuation Shadow: Another phenomenon which has not been considered by the Applicant in establishing its EPZ is the "evacuation shadow," or "tendency of an official evacuation advisory to cause departure from a much large area than was originally intended." Ziegler, et al, "Evacuation from a Nuclear Technological Disaster", Geographical Review, supra, at 7. At Three Mile Island, where pregnant women and preschool children within a radius of five miles were advised to evacuate and everyone within a ten-mile radius was advised to remain indoors, 39% of the population living within 15 miles of the plant evacuated (53% within 12 miles). Between 10 and 12 miles, some 47% of households sampled evacuated some of their members. Geographical Review at 6-7. An even greater response could be expected following an evacuation directive in an area with a radius of 10 miles or more. Such a response would have a significant impact on the Applicant's ability to plan for an emergency, and thus on the size and dimension of the EPZ. Consideration of the evacuation shadow, as based on existing studies of the phenomenon and the population characteristics of the area surrounding the Seabrook facility, should be included in the Applicant's determination of the proper size and shape of the EPZ.

6. The emergency plan does not indicate that beyond design basis accidents were considered in the establishment of the EPZ.

Thus, the Applicant has not planned for the type of accident involving the greatest risk to the public. The failure to consider such an accident as part of the EPZ is generally justified by reference to the low probability of a core meltdown determined in NUREG-0396. However, the foundation for this determination, WASH-1400, has been largely discredited. As stated by the NRC's own Risk Assessment Review Group Report, NUREG/CR-400 at 25, "when a great many rare events each have a small probability occurrence, the chance that at least one of them will occur can be rather high." In any event, NUREG-0396's determination of a 30% probability that if a core meltdown occurs it will result in doses above the Protective Action Guidelines constitutes sufficient justification for its consideration in establishing the EPZ. The need to expand the EPZ relates both to improving emergency planning responses and limiting health effects of a major accident. As the government of California's Office of Emergency Services, which has adopted expanded EPZ's in consideration of beyond design basis accidents, has stated:

> "[P]rudence dictates that the EPZ's be extended so that advance planning can be performed to aid in resolving the potential problems associated with the more severe types of accidents, the penetration leakage and major containment failure classifications...It did not seem prudent to restrict planning attention to responding <u>only</u> to the potential for incurring immediate life threatening radiation doses for such severe accidents. Thus the extended zones were selected so that the potential for incurring health impacting doses was reduced not only for early fatalities, but also for early injuries and delayed cancer effects as well...[E]xtending the EPZ boundaries results in a prudent reduction in the probabilities of early health effects and a substantial reduction in the probability of delayed health effects (associated with) .5 to 1 Rem PAG dose limits."
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> * * *
>
> "Not only does the extension [of the Plume EPZ] improve the probability of limiting doses to the public, but it

> also substantially improves the base of procedural plans and facilities for required emergency response efforts with respect to evacuation, sheltering and relocation preparedness and capacity requirements."

State of California, Office of Emergency Services, "Emergency Planning Zones for Serious Nuclear Power Plant Accidents", November 1980, Alex R. Cunningham, Director.

7. The Applicant does not provide offsite emergency plans of state and local governments, nor does it indicate how the Applicant's emergency plan will coordinate with those offsite plans. 10 CFR 50.33(g), Appendix E.III. The plan should show that all signals emanating from the plant receive the same interpretation by state and local authorities as they are given by the utility. In addition, letters of agreement between the Applicant and the affected state and local governments, reflecting their mutual understanding of each other's responsibilities, must be submitted to the NRC. NUREG-0654, App. 3. NECNP reserves the right to amend its contention to challenge the sufficiency of the Applicant's plans to coordinate with state and local authorities.

8. The FSAR does not show that all possible accident sequences can be monitored, as required in order to take the necessary steps in responding to an emergency. For example, there is no indication that process monitors comply with General Design Criteria 13, 19, and 64, as implemented by Reg. Guide 1.97, or are environmentally qualified as required by GDC 4.

9. The FSAR does not provide for adequate radiological monitoring. Previous experience has shown that reliance upon individuals to take radiation samples in the field may lead to

serious errors. See NUREG-0600, at II-3-71 - -7, -97. Because weather conditions may impede utility personnel and human error factors may detract from the accuracy of results, permanent radiological monitoring equipment should be placed at a number of locations surrounding the plant. These monitors should relay information to a computer at the plant which can plot radiation levels and estimate the location of the plume. The monitoring equipment should be equipped with independent backup power supply and must meet criteria for withstanding adverse meteorological conditions.

10. The Applicant must submit and justify a dose assessment model. App. E to Part 50. This model must not be limited to a straight diffusion model, but must take into account the behavior of heated releases and other "source characteristics." NUREG-0654, App. 2 at 2-3. The computer used for making dose assessments should have an independent backup power source to assure that it will continue to operate if onsite or offsite power fails.

11. The emergency plan does not provide for early notification and clear instructions to the local populace, as required by 10 CFR 50.47(a)(5). Detailed criteria for providing notification to the public, set forth in NUREG-0654 App. 3, must be met before the Applicant may be licensed. The Applicant must provide a means by which all people within the EPZ will be able to hear a warning of a radiological emergency, and they must be trained to understand the warning. In the Seabrook area, special provision must be made for educating the thousands of transients who move through the

area in the summer months. In particular, they should be apprised in advance of the location of shelters evacuation routes, and congregation areas.

12. The emergency plan does not provide for the sheltering of the large numbers of people who may be on beaches during a radiological emergency in the summer, and who will not be close to their own homes, motels, or public buildings.

13. The emergency plan does not indicate the basis for the code it uses to make evacuation time estimates. The plan does not indicate its bounds of error, or whether the model is static or dynamic. A sensitivity analysis should be performed for the model and for its underlying assumptions, and the Applicant should be required to disclose those assumptions which undergird the evacuation time estimates. Furthermore, the Applicant should indicate its reasons for substantially reducing the estimates of evacuation time presented in its PSAR. NECNP reserves the right to amend its contention to challenge the bases for the Applicant's evacuation time estimates and their accuracy when this information is submitted.

14. The preliminary evacuation time estimates submitted by the Applicant assume conditions much more favorable to rapid evacuation than are likely to occur. At the very least, the Applicant should consider a "worst case" scenario in order to assess the correctness of its estimates and assure that emergency planning is adequate to meet these circumstances. For example, the combination of weather and population conditions assessed by the ETAs did not include adverse weather conditions on a summer weekend, when the Seabrook population swells to its greates number.

Second, the ETAs assume that normal traffic patterns will prevail. The ETAs should instead take into account the evacuee directional bias discussed in these contentions at ¶4, and the possibility that much of the traffic will move upwind of the plant. Third, the ETAs should take into account the delays caused by substantial evacuation from the "evacuation shadow." Fourth, the ETA estimates were based on evacuation by private vehicle, which excludes the more time consuming and complex factor of the use of buses to evacuate children in schools, children in camps, tour groups travelling in buses, and people in institutions such as hospitals and nursing homes. In its present form, the Applicant's preliminary evacuation time assessment is far too simplistic. NECNP reserves the right to challenge the adequacy of the Applicant's ETAs in their final form.

15. The Applicant has not satisfactorily demonstrated the feasibility of evacuating the Seabrook EPZ without exposing evacuees to unacceptable radiation doses. The Commonwealth of Massachusetts has calculated that, under conditions postulated by the NRC/EPA Task Force Report (which has been endorsed by the Commission), a radioactive plume travelling at a speed of 10 mph over the two miles between the beach and the plant could reach Hampton Beach within 15 minutes of a release. Furthermore, a plume travelling to Salisbury Beach at the same speed could reach the beach in about 30 minutes. Commonwealth of Massachusetts Memorandum in Support of SAPL's Request for an Order to Show Cause (May 2, 1979) at 11, citing NRC/EPA Task Force Report, NUREG-0396 at 20, App. I. The emergency plan's evacuation time estimate for Hampton Beach is

4 hours and 20 minutes, and for Salisbury, 3 hours and 50 minutes. As discussed above, NECNP considers these estimates to be extremely optomistic.

Even if the Applicant could show that it had supplied enough shelters in the area for the large numbers of people who might be exposed on the beach or otherwise outdoors on a summer day, the sufficiency of shelters to protect against atmospheric radiation is in grave doubt. See NUREG/CR-1131, "Examination of Offsite Radiological Emergency Measures for Nuclear Reactor Accidents Involving Core Melt, Sandia Laboratories, June, 1978. ("Sheltering by itself, unless the quantity of radionuclides inhaled can be substantially reduced, will also not provide much protection." Id. at 92.)

The Applicant should therefore be refused a license because it has not provided a reasonable assurance that its emergency plan can and will protect the public from excessive exposure to radiation.

16. In order to plan adequately for the protection of the public health and safety, baseline data on local health conditions should be gathered to determine the immediate and long-term health effects of radiation exposure.

> Data to be collected should include at least measurements of thyroid hormone deficiency in newborns, fetal death rates, neonatal and infant morbidity and mortality rates, known exposure to carcinogens, cancer incidence and prevalence withing circumscribed areas around nuclear facilities, occupational history, demographic characteristics, and symptoms of psychological distress of the population at risk, as well as the availability of medical staff support and health care facilities during a radiological emergency.

Gordon MacLeod, M.D., "A Role for Public Health in the Nuclear Age," American Journal of Public Health, Vol. 72 No. 3 (March 1982.) The Applicant has not shown that the health effects of a radiological emergency can or will be monitored in the Seabrook area. It should be required to do so before a license is issued.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, et al.,	)	Docket Nos.
(Seabrook Station Units 1 and 2)	)	50-433 OL 50-444 OL

CERTIFICATE OF SERVICE

I hereby certify that copies of the NECNP SUPPLEMENTAL PETITION FOR LEAVE TO INTERVENE in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, this 21st day of April, 1982.*

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